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Comparative validation of Monte Carlo codes for the conversion of a research reactor

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ABSTRACT

This paper presents the calculation results of the set of test problems for a research reactor with a tube-type low enriched uranium (LEU, 19.7 w/o, U-9%Mo) fuel and oxide high enriched uranium (HEU, 90 w/o) fuel, a light water moderator, and a beryllium reflector. The static cases and the depletion problem were examined. Calculations were performed using continuous energy Monte Carlo codes: MCNP (+MCREB for burnup calculation), MCU-PTR, and SERPENT 2. The impact of the cross-section libraries used for a particular problem on the calculated results was investigated.

1. Introduction

Feasibility studies regarding the conversion of many research reactors to low enriched uranium (LEU) fuel are currently performed worldwide. In many cases, these studies stimulate the development and validation of research reactor core models and codes. Usually, the validation of neutronics calculation codes is based on the comparison either with criticality safety benchmark results (e.g., the International Criticality Safety Benchmark Evaluation Project (ICSBEP) (OECD/NEA, 2009)) or with the reactor plant operational data. However, there are still few evaluated benchmark results available for heterogeneous thermal systems with LEU (20 w/o) fuel, and the research reactor configuration is usually more heterogeneous compared to that found in critical assembly facilities (Liem and Sembiring, 2012). No reactor experimental data exist for the new types of LEU fuel currently under development. Therefore, for the validation of the calculation models of the cores using the new types of fuel, the procedure of comparative validation of precision Monte Carlo codes was proposed. The comparison of the calculations conducted by Monte Carlo codes with different cross-section libraries is an important instrument for the validation of codes in the case when the experimental data are also available (Liem and Sembiring, 2012; Savva et al., 2014).

The present work initially was performed for the conversion analysis of the IRT MEPhI research reactor at the National Research Nuclear University (NRNU) MEPhI. The IRT MEPhI is a 2.5-MW pool-type research reactor. Currently, IRT-3M high enriched uranium (HEU, 90 w/o) fuel assemblies are used. The feasibility studies were completed in November 2011. For the feasibility studies of the IRT MEPhI reactor conversion, the tube-type fuel assembly IRT-3M with U9%Mo-Al fuel (19.7 w/o) was chosen as the LEU fuel (Izhutov et al., 2008).

In the framework of the conversion feasibility studies, the detailed neutronics core analysis using the Monte Carlo codes MCNP (X-5 Monte Carlo Team, 2003) (using MCREB Stevens, 2008 for the burnup calculation) and MCU-PTR (Alekseev et al., 2011, 2012) was performed. To validate the obtained results, a comparison was performed of the calculations using these codes on some test problems. The test problems were developed for the IRT-type reactor with U-Mo LEU fuel and oxide HEU. The real geometry of the fuel assembly was considered. The core configuration was simplified: the experimental channels, the end details of the fuel assemblies, the grid plate, among other features were not considered. The static cases and the depletion problem were examined.

The calculations of the proposed test problems can be used not only in the framework of the conversion analysis of a particular reactor but also for the validation of different codes. For example,

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the calculations of test problems were performed for the validation of SERPENT 2 (Leppänen, 2005, 2010) code, which is used for the SAFARI reactor calculation.

2. Models and codes

2.1. MCU-PTR

The MCU-PTR code (Alekseev et al., 2011) was used for steady-state neutronics calculation and for the burnup calculation. MCU-PTR is the code for the calculations of pool- and tank-type research reactors in the framework of the MCU-5 package. The MCU-5 package (Alekseev et al., 2012) is intended for simulation of the transport of neutrons, photons, electrons, and positrons using analogous and weighted Monte Carlo methods. The MCU-5 package is used for the calculation of the different types of power and research reactors. The data bank of the MCU-5 package (MCUDB50) includes Russian libraries and the available foreign libraries. The MCUDB50 data bank contains the data for 375 isotopes.

According to Alekseev et al. (2012), when simulating the collisions of neutrons with nuclei in different energy regions, it is possible to combine different data libraries. In the fast energy region, e.g., it is possible to use either the constants of the ACE/MCU library or the BNAB/MCU library. ACE/MCU is the library of cross sections of neutron interaction with nuclei in the epithermal energy region in a pointwise representation obtained from the ENDF/B-VII.0 files and other sources. BNAB/MCU is an expanded and modified version of the BNAB-93 26-group system of constants. The energy boundaries of the regions for the use of ACE/MCU and BNAB/MCU are defined by the code user.

In the thermalization region, it is possible to use a multigroup transport approximation or to perform a simulation using continuous-energy neutron interaction data.

The MCU-5 package has a burnup module that is intended for calculating the nuclide composition of the reactor materials in the process of reactor operation. The equations of isotopic composition kinetics are solved using an iterative (predictor corrector) method.

In the present work, two variants of MCU-PTR calculations were examined:

- using the BNAB/MCU library for the energy region $E > 4.65$ eV (the ACE/MCU library is not used);
- using the ACE/MCU library for the energy region 100 keV $< E <$ 20 MeV and the BNAB/MCU library for the energy region 2.15 eV $< E <$ 100 keV).

For the energy region $E < 2.15$ eV or $E < 4.65$ eV in both variants, the same continuous-energy neutron interaction data were used.

The MCU-PTR calculations were performed at NRNU MEPhI.

2.2. MCNP and MCREB

The MCNP code (X-5 Monte Carlo Team, 2003) with the ENDF/B-VII.0-based cross-section libraries was used for the steady-state neutronics calculations of the test problems. The calculations with previous versions of ENDF/B were also performed.

The GTRI program at Argonne National Laboratory (ANL) developed a relatively easy method to perform Monte Carlo burnup analyses (Hanan et al., 1998). The Monte Carlo burnup analyses are performed using the MCREB code. This code uses the MCNP code for the calculation of k_{eff}, neutron fluxes, and cross-sections. These one-group neutron fluxes and cross sections (capture, fission, n-2n, n-α, and n-p) are then supplied to the REBUS (Olson, 2001) code for the power normalization and calculation of the burnup-buildup of the relevant isotopes. The REBUS code then supplies the "burned" compositions for the MCNP code for the next step in the analysis.

The MCNP calculations with the ENDF/B-VII.0-based cross-section libraries and the calculations using the MCREB code were performed at ANL.

The MCNP calculations with the ENDF/B-VI and ENDF/B-V-based cross-section libraries were performed at NRNU MEPhI.

2.3. SERPENT 2

The Monte Carlo reactor physics code SERPENT 2 (Leppänen, 2005, 2010) with the ENDF/B-VII-based cross-section libraries was used for the steady-state neutronics calculation and for the purpose of simulating depletion of the core. SERPENT 2 uses continuous energy ACE format data libraries. With the exception of the thermal cross sections for beryllium, all libraries used in this study were based on the ENDF/B-VII.0 evaluation. For the depletion studies, ENDF/B-VII.0-based fission yield and decay libraries were also used. Depletion was performed with the built-in depletion module utilizing the predictor–corrector method.

The SERPENT 2 calculations were performed at Necsa.

3. Test problem description

The cross-sectional view of the IRT-3M fuel assembly (FA) with LEU fuel is shown in Fig. 1. The 6-tube FA consists of 6 co-axial fuel tubes with a control rod (CR) channel in the center. The dimensions of the 6-tube FA are presented in Table 1. The outer and inner dimensions of the fuel tube ($S1$ and $S2$, respectively) and the outer and inner radii of the rounded corners (R and r, respectively) are presented. The first tube is the outer tube.

The outer dimensions of the IRT-3M FA with HEU fuel are the same as the dimensions of the FA with LEU fuel, except for the radii of the rounded corners ($R = r = 0.4$ cm for all tubes). Table 2 presents the data regarding the 6-tube FA IRT-3M with LEU and HEU fuel used in the study.

Table 3 presents the general description of the reactor design parameters for the test problems.

The core consists of 48 cells (6×8 positions) for the FA and reflector blocks. There is water between FA (reflector blocks) and in the cells without FA or reflector blocks. The core height is 58 cm. The water reflector thickness is 3×7.15 cm in the X–Y direction and 29 cm in the axial direction. The vacuum boundary condition (BC) is used at the external border. One-half of the described system in the axial direction is considered for all of the test problems. Additionally, for the test depletion problem, one-half of the described system in the horizontal plane is considered. Reflection boundary conditions are used at the symmetry axes. The

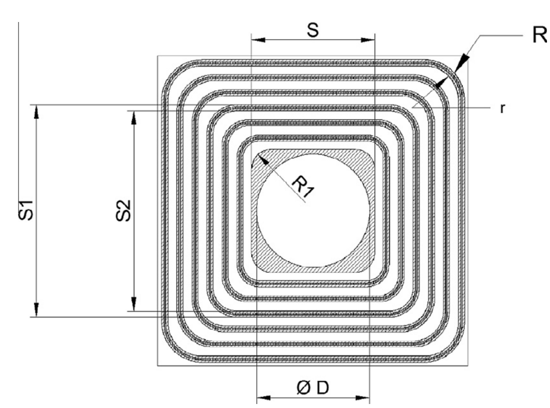

Fig. 1. FA geometry.